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Interim Report – Aging Effects of Important Balance of Plant Systems in Nuclear Power Plants

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EXECUTIVE SUMMARY

In any commercial nuclear power plant, there are a substantial number of Balance of Plant (BOP) systems besides the commonly thought of main steam, feedwater, turbine-generator, condensate, etc.In recent years, BOP systems have arisen as major causes of plant transients, e.g. the June 9, 1985 Loss of All Feedwater event at Davis-Besse, and have received increased attention from the nuclear industry and the NRC.

This is an interim report describing the activities to date by Brookhaven National Laboratory (BNL) to support the Nuclear Plant Aging Research (NPAR) Program's study of Balance of Plant (BOP) systems. This initial phase of the study provides a rationale for and a preliminary indication of those BOP systems which may warrant detailed study for the effects of aging. A brief discussion of the suggested approach to accomplish the overall objective of identifiying the effects of aging of these BOP systems on nuclear plant safety is also presented.

In this study, BOP systems have been defined as all nonsafety-related systems, except for those associated with the Nuclear Steam Supply Systems (NSSS) such as Control Rod Drive, Pressurizer Power Operated Relief Valves (PORVs) and discharge piping, Pressurizer Relief Tank (PRT), etc. for PWRs and Reactor Core Isolation Cooling (RCIC), Control Rod Drive Hydraulic (CRDH), etc. for BWRs. Some of the nonsafety-related NSSS are being studied in other phases of the NPAR program.

From the results of the study, it was concluded that the frequency of unplanned reactor scrams has been more frequently cited as an indicator of safety performance and that the frequent contributors to unplanned reactor scrams caused by BOP systems are largely the power conversion systems, i.e. the feedwater, main turbine, main generator, main steam (usually the steam bypass to the main condenser) and the condensate systems. Other frequently contributing BOP systems are support systems such as the Electrical Distribution System, and less frequently the Circulating Water, Service/Instrument Air, Fire protection and HVAC Systems. The Electrical Distribution System consists of the 120V AC Power, switchyard, large plant loads, DC power and control centers. At a component level, the feedwater regulating valves and turbine-driven feedwater pumps are frequent conributors, as well as the main turbine electro-hydraulic control (EHC) subsystem. Failures in the main generators are also important.

These results coincide substantially with the results from an alternative approach in which important BOP systems were categorized considering PRA insights.

Preliminary recommendations are as follows:

- The frequency of unplanned reactor scrams be considered the most important indicator of current, or potentially near term, safety problems.
- BOP systems that should be included within the NPAR program are those which significantly contribute to unplanned reactor scrams.

The next phase of this program will focus on several of the oldest plants. LERs involving unplanned reactor scrams since the beginning of commercial operation until the present time will be reviewed to determine if there is an increasing frequency of unplanned scrams caused by the identified BOP systems. This group will include Monticello and Yankee Rowe, the pilot plants for the plant life extension study.

After completing the above, a group of intermediate age plants will be examined in the same manner, followed by the youngest plants. These three groups will be compared and analyzed. The ultimate goal is to determine whether aging of the individual system components is a significant factor affecting nuclear plant safety. This will be done in the same manner as has been done for safety-related systems within the NPAR Program.

1. INTRODUCTION

The U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research is the prime sponsor of the ongoing Nuclear Plant Aging Research (NPAR) Program which focuses on the long-term safety implications of aging of nuclear plant systems and components. In recent years, Balance of Plant (BOP) systems have arisen as major causes of plant transients, e.g. the June 9, 1985 Loss of All Feedwater event at Davis-Besse, and have received increased attention from the nuclear industry and the NRC. As such, these systems have also been selected as targets for study in the NPAR Program.

1.1 Purpose

In any commercial nuclear power plant, there are a substantial number of BOP systems besides the commonly thought of main steam, feedwater, turbine-generator, condensate, etc. This is an interim report describing the activities to date by Brookhaven National Laboratory (BNL) to support the NPAR Program's study of BOP systems. This initial phase of the study provides a rationale for, and a preliminary indication of, those BOP systems which may warrant detailed study for the effects of aging. A brief discussion of the suggested approach to accomplish the overall objective of identifiying the effects of aging of these BOP systems on nuclear plant safety is also presented.

In this study, BOP systems have been defined as all nonsafety-related systems, except for those associated with the Nuclear Steam Supply Systems (NSSS) such as Control Rod Drive, Pressurizer Power Operated Relief Valves (PORVs) and discharge piping, Pressurizer Relief Tank (PRT), etc. for PWRs and Reactor Core Isolation Cooling (RCIC), Control Rod Drive Hydraulic (CRDH), etc. for BWRs. Some of the nonsafety-related NSSS are being studied in other phases of the NPAR program.

1.2 Study Approach/Scope

In Section 2 of this report, a summary of prior NRC-sponsored BOP-related activities and findings is presented. This information is necessary to ensure that a consensus definition of BOP systems is being applied in this study and is also very useful in identifying and prioritizing the important BOP systems.

These activities include the following:

- Environmental qualification of Class IE Equipment.
- AEOD studies of engineered safety feature (ESF) actuations and unplanned reactor scrams.
- NRR-sponsored study by the MITRE Corporation concerning regulatory considerations for Balance of Plant.
- June 9, 1985 Davis Besse Event – NRC Lessons learned
- SECY-86-349 Policy Issue for Balance of Plant
- NRC inspection procedures for Balance of Plant, including the temporary instruction for maintenance team inspections (MTI).
- Recent AEOD performance indicator – related efforts such as NUREG-1275, operating experience feedback reports for new plants and also progess in scram reduction, as well as the ongoing development of a maintenance effectiveness indicator (MEI).
- NRR survey of plant design strengths and weaknesses.

Nearly all of the above-referenced documents particularly SECY-86-349, the staff policy on BOP, convey the emphasis placed by the NRC to reduce the frequency of challenges to plant safety systems. In turn, unplanned reactor scrams are the leading contributors to challanges to safety systems, so that extensive NRC focus has been placed on reducing their frequency.

In Section 3, an assessment is made of the important BOP systems based on their contribution to unplanned reactor scrams. An alternative approach in which BOP systems are categorized by expert opinion considering insights from probabilistic risk assessments (PRA) is also presented for comparison purposes. In addition, the relevant aspects of a prior study by

BNL staff regarding reactor trips and their relationship to initiating events appearing in a PRA is also presented. The study proposed utilizing the frequency of abnormal events as a risk-based performance indicator.

2. SUMMARY OF PRIOR NRC-SPONSORED BOP ACTIVITIES

2.1 Environmental Qualification of Class IE Equipment

With the publication of IE Bulletin 79-01, "Environmental Qualification of Class IE Equipment" on February 8, 1979, and subsequent revision through Bulletin 79-01B, the NRC established certain requirements for the environmental qualification of nonsafety-related equipment important to safety. These requirements were formally established in 10CFR50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants" as follows:

"(b) Electric Equipment important to safety covered by this section is:

(1) .....

(2) Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (i) through (iii) of paragraph (b)(1) of this section by the safety-related equipment."

According to paragraph (b)(1), safety-related electric equipment is that relied upon to remain functional during and following design basis events to ensure:

(i) the integrity of the reactor coolant pressure boundary,

(ii) the capability to shutdown the reactor and maintain it in a safe shutdown condition, and

(iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to 10CFR100 quidelines.

Licensees were also required in paragraph (d) to prepare a list of equipment important to safety required by Section 50.49.

Some licensees responded by separating the equipment into two categories: Balance of Plant (Inside and Outside Containment) and NSSS (Inside Containment).[1] However, this definition appears to include primarily the traditional non-NSSS safety-related components as BOP components.

It should be noted that aging of BOP components is indirectly considered in environmental qualification as a significant cause of the postulated environmental conditions to which the safety-related components would be subjected. That is, stress-related aging, corrosion or erosion of so-called high energy and moderate energy piping fluid systems both inside and outside containment leading to rupture or cracking of the piping would in turn cause safety-related components to be subjected to pipe whip, jet impingement, moisture and/or flooding conditions.

2.2 AEOD Studies of Engineered Safety Feature Actuations and Unplanned Reactor Trips

2.2.1 Engineered Safety Feature Actuations

In 1985 and 1986, the NRC's Office for the Analysis and Evaluation of Operational Data (AEOD) issued studies of engineered safety feature (ESF) actuations at nuclear power plants for the periods of January 1 through June 30, 1984 and July 1 through December 31, 1984, respectively.[2,3] In general, the BOP systems involved in the actuations which occurred were HVAC systems for the control building due to overly conservative radiation monitor settings or actual toxic gas infiltration into the control room at a limited number of plants. For the most part, strictly defined nonsafety-related systems were not significant contributors to ESF actuations.

2.2.2 Unplanned Reactor Trips

Also in 1985 and 1986, AEOD issued trends and patterns reports of unplanned reactor trips which occurred in 1984 and 1985, respectively.[4,5] These studies provided the first clear indication of the significant impact of BOP systems and components on challenges to safety systems, e.g. unplanned reactor trips.

In both years, the majority of reactor trips occurred with the reactor power above 15%, in 1984 – 68% and in 1985 – 75%. In 1984 and 1985, 31% and 38%, respectively, of all trips occurred while the plant was at 95% power or above.

- Above 15% power, the major BOP systems, collectively referred to as the Power Conversion Systems, i.e. feedwater, turbine, condensate, main steam and main generator contributed to 59% and 52% of all reactor trips, in 1984 and 1985, respectively.
- Hardware failures were the dominant cause of unplanned reactor trips above 15% power. In 1984 and 1985, hardware failure contributed 60% and 55% respectively of all trips above 15% power. The Feedwater Regulating Valves, Main Steam Isolation Valves, Turbine Control Valves, and Turbine Stop Valves were major contributors. These valves were responsible for 8% of all trips above 15% power in 1985.
- The Electrical Distribution and Reactor Protection Systems were the major non-power conversion system contributors in both 1984 and 1985, contributing 23% and 29% respectively of trips above 15% power. (Portions of the Electrical Distribution System can be considered BOP since they are nonsafety-related.)

2.3 NRR Sponsored Study by the MITRE Corporation Concerning Regulatory Considerations for Balance-of-Plant

In October 1986, NUREG/CR-4783 was issued which is the final report by the MITRE Corporation concerning certain regulatory and performance aspects of the conventional or power conversion side of nuclear power plants, also referred to as the Balance-of-Plant.[6] This study was commissioned by the Office of Nuclear Reactor Regulation and essentially concurs with the previously described AEOD results by focusing specifically on BOP systems only and providing a more detailed analysis separating the data into different categories based on reactor type (PWR or BWR) and age, amongst others.

In the report, MITRE characterizes and analyzes BOP failures and provides a perspective on the significance of the failures relative to overall plant safety. The report also encompasses recommendations to enhance the NRC regulatory approach to BOP performance and reviews current BOP-related activities and identified gaps. However, the report was not intended to focus exclusively, if at all, on the effects of aging of BOP systems. Many of the failures identified pertained to operator errors.

The findings for the periods studied, i.e. 1984 and 1985, concur with the previous AEOD studies in that for PWRs, condensate and feedwater accounted for 52% of BOP-caused reactor trips while 30% were due to the turbogenerator. Similarly, for BWRs, 47% of trips were caused by condensate and feedwater, and 26% were attributable to the turbogenerator.

Other significant conclusions were that:

- BOP-related trips constitute 70% of all plant trips (for the time period studied).
- There are large unit-to-unit variations in BOP performance: 60% of the mean BOP trip frequency among older plants and over 100% among newer ones.
- Units that have above-average performance in the initial years tend to stay better, and vice versa.
- The plants which were outliers amongst the older unit group account for about twice their proportionate share of BOP-related trips.
- Personnel errors account for about 50% of the known causes of BOP trips.
- About 40% of the BOP trips due to human error occur during maintenance, test, repair and inspection activities.
- Root causes in more than 33% of plant component failures are listed as unknown.
- Single component failures within the turbogenerator, condensate and related support systems account for about 50% of BOP trips.

2.4 June 9, 1985 Davis-Besse Loss of All Feedwater Event – NRC Lessons Learned

On November 26, 1985, William J. Dircks, the Executive Director for Operations, issued an internal NRC memorandum concerning the Lessons Learned from the June 9, 1985 loss of all feedwater event at Davis-Besse.[7] The first lesson concerned more timely identification and completion of safety issues, the second concerned broader consideration of positive and potentially negative safety impacts of regulatory actions, while the third reads as follows:

> "3. Increased Emphasis on Balance of Plant Equipment
>
> The paramount importance of proper maintenance in maintaining levels of reliability assumed in the Safety analyses that form the licensing basis for operating plants has been accorded greater recognition and increased emphasis and attention by both NRC and utility management in the aftermath of the TMI accident. However, it appears from the circumstances noted in the review of the June 9 Davis-Besse event that an inappropriate, artificial distinction (alluded to in 2. above) between the importance of safety-related vs nonsafety-related plant features may have led some licensees to place inadequate emphasis on proper maintenance of all equipment necessary to assure proper facility operations. Some balance-of-plant systems may actually have equal or perhaps greater safety importance (cumulatively) than equipment classified as safety-related because their too-frequent failure can needlessly challenge the safety-related systems, and their failure can also aggravate conditions under which the safety-related systems must respond. We need to give increased attention to assuring that the attention of licensee management is focused properly on this important aspect of plant operations, and that important balance-of-plant systems and equipment receive adequate attention in the overall maintenance picture. We should

also consider seriously, in the context of finalizing our improvement plans, whether this requires significantly increased commitment of regulatory attention to balance-of-plant areas within our licensing review and inspection program."

2.5 SECY-86-349 Policy Issue for Balance Plant

As a result of the Davis-Besse memo and the concurrent AEOD and NRR reports, the NRC staff provided to the Commissioners an Information Policy Issue for Balance of Plant, designated SECY-86-349, dated November 21, 1986.[8] This SECY document was intended to describe the BOP programs currently underway and those planned for the future.

With respect to the inspection process, the document states that licensee activities relating to the design, maintenance, operation and testing of BOP systems and equipment may be examined to the degree they impact on safe operations.

Furthermore, it states that inspections of steam and power conversion BOP systems are mostly reactive in nature. That is, where there are trips and transients or recurring performance problems with BOP systems and equipment which present significant challenges to safety systems, inspectors will, to varying degrees, explore causes and corrective actions with the licensee. For example, post-trip reviews may lead to investigations and evaluations of BOP where reactor trips are initiated by BOP equipment failures. Where events such as feedwater pump trips recur and the licensee is not identifying and dealing with the underlying cause, the inspection process may result in discussions with the licensee and requests for appropriate licensee actions.

The document summarizes some of the specific approaches being taken by the Regional Offices in addition to the reactive BOP inspections, as follows:

- Some Regional Offices address BOP performance and its impact on operational performance in SALP reports. In one case, statistics have been kept on several important performance indicators such as the number of reactor scrams initiated by BOP equipment. Licensee BOP performance is assessed in terms of this information and underlying problems are identified where they are known.

- Inspections are performed on modifications and testing of certain selected and reviewed non-safety-related or BOP modifications which are judged to rank high in importance to overall plant safety.
- Resident Inspectors in the normal course of their plant walkthroughs observe material conditions, housekeeping and work ongoing in BOP areas. Where problems are observed, these are brought to the licensee's attention.
- In light of available information from PRA studies, IE[‡] and the Regions are developing trial programs for inspections to focus more on major risk contributors irrespective of safety classification. In some cases this is significant in that BOP systems have interconnections with safety-related systems such that their availability and/or reliability is affected by the BOP portion. Consequently, the staff is developing tabulations from selected existing PRAs that would identify risk-significant systems and components, associated failure mode(s), and identify areas warranting more frequent system walkdowns. Existing PRAs are being utilized to develop information which can be applied to similar plants not covered by a plant-specific PRA.

Regarding future plans, the document mentions that a trial inspection program of a selected important BOP system such as the feedwater system is being developed. This is discussed in the next section.

[‡] Since incorporated into NRR.

2.6 NRC Inspection Procedures for Balance of Plant

2.6.1 Temporary Instruction 2515/83 Balance of Plant Trial Inspection Program (Feedwater System)

As a result of the preceding activities concerning BOP, and as stated in SECY-86-349, the NRC issued Temporary Instruction 2515/83 effective February 26, 1987.[9] The purpose of the instruction was to implement a limited trial program to inspect a generic BOP system, i.e. the Feedwater System, that has been shown to be a significant contributor to unplanned challenges to reactor safety systems, as well as a significant contributor to overall plant risk from a probabilistic risk assessment (PRA) perspective. The results of the trial inspection program were used to assess how inspections of BOP equipment can be effectively incorporated into the routine inspection program and to assist in determining whether greater emphasis should be placed on inspecting BOP systems. The basic guidance specified in the procedure is to assess licensee responsiveness to and adequacy of corrective action as an indicator of the licensee's commitment to safety. The general categories for which specific inspection instructions are given are operations, maintenance, modifications, design, and management support.

Aging of components is not directly cited, but rather as an indirect cause of equipment failure which should be evaluated for input to the preventive maintenance program and for possible design modifications. The corrective maintenance program is inspected to ensure that corrective maintenance is documented in maintenance history records, appropriately trended, analyzed, and applied to the preventive maintenance program through procedural revisions or design changes.

Under operations, the inspection covers general conditions of equipment, e.g. no excess corrosion, no evidence of steam, water or oil leaks, as well as verification that Feedwater System corrosion control measures are in place.

It would appear that aging of components would not ordinarily be directly cited as a cause of feedwater equipment failure by the NRC inspectors.

2.6.2 NRC Inspection procedure 71500 – Balance of Plant Inspection

Upon considering the results of the temporary instruction discussed above, the NRC determined the need exists for a permanent inspection procedure for BOP, and which was not limited to the Feedwater System. Such a procedure was issued as No. 71500 on September 30, 1988.[10]

Inspection Procedure 71500 differs from Temporary Instruction 2515/83 in the following ways: Design is no longer a separate category but has been combined with modifications as a single category. Also, a new category has been added entitled Root Cause.

The procedure relates only indirectly to aging of BOP components, in the same manner as the previously described temporary instruction, for maintenance and operations. A more direct relationship to aging of components may result from the procedure's instructions for Root Cause. The inspection guidance states that at least four components that have a history of unreliability or have caused or complicated recovery from plant trips or transients should be selected. The root cause for each component failure or event should be identified by reviewing the licensee's root cause analysis, if performed. The analysis should implicate broad programmatic areas such as maintenance, operator training, etc. that are contributors to the component failure or event, and that are in need of upgrade.

2.6.3 Temporary Instruction 2515/97 Maintenance Inspection

In the last few years, the NRC has intensified its emphasis on maintenance, most importantly by publication in the Federal Register of proposed rules to ensure the effectiveness of maintenance programs for nuclear power plants on November 28, 1988.[11] The proposed rule states that it is the objective of the Commission that all components, systems and structures of nuclear power plants be effectively maintained so that plant equipment will perform its intended function when required. The scope of the proposed rule is intended to cover all systems, structures and components including those in the Balance of Plant (BOP).

To provide the background to the proposed rule, the NRC has had in effect a Temporary Instruction, 2515/97, titled "Maintenance Inspection."[12] It states that the inspection team will determine what failures of significant equipment (probabilistic risk assessment (PRA)-identified, safety-related, or BOP that affects safety-related and special interest items) have occurred and will inspect the licensee's trending and maintenance activities to schedule, repair, and prevent further failure of that equipment. Other selective examinations of equipment failures attributed to maintenance will be examined to determine the adequacy of licensee corrective actions and root-cause determinations of the failures.

By means of the temporary instruction, the NRC has been conducting maintenance team inspection which will cover all U.S. nuclear power plants and are targeted for completion by April 1991. It appears that the inspection reports arising from this effort may provide valuable insights into aging of BOP components.

2.7 Recent AEOD Performance Indicator – Related Efforts

In the years since the first AEOD studies of Engineered Safety Feature Actuations and Unplanned Reactor Trips, as described previously in Paragraphs 2.2.1 and 2.2.2, efforts have been directed generally within the NRC to establish various indicators of plant performance. Particular emphasis has been placed on reductions in the frequency of unplanned reactor trips (scrams), and also on the effectiveness of utility maintenance programs as a means to reduce the scram frequency. AEOD's efforts in these areas are described below.

2.7.1 Operating Experience Feedback Report – New Plants (NUREG-1275, Volume 1)[13]

In this report, mature plants are defined as those which were licensed before January 1, 1983 (76 plants). The study scope included the operational experience for 22 new plants from January 1983 through June 1986. The data for mature plants, which were examined for contrast, were obtained from licensee event reports (LERs) submitted for Calander Year 1985. With respect to BOP, the report concluded that:

- On a percentage basis, the causes of scrams at new plants are very similar to those of mature plants. The primary causes are associated with the BOP systems, with the Feedwater System dominating.
- Focusing on the BOP prior to operation and early in plant life appears to provide a high return regarding the reduction of unplanned scrams and ESF actuations.
- In particular, additional reviews of feedwater and turbine control and bypass systems should be conducted to identify sensitivities and plant-specific characteristics that could contribute to transients, or the (in)ability of the system to cope with expected transients.
- Conducting systematic reviews of equipment-protective logics and setpoints on components such as pumps (suction trip, time delay, vibration trip) or power supplies to identify areas where a time delay or additional channels for coincidence could reduce the potential for unnecessary transients or spurious actuations. Special attention should be paid to first-of-a-kind features not incorporated in earlier designs, such as the main steam reheater drain high level trip and other turbine protective trips.
- More focused attention should be given to BOP operations, e.g. the ability of the operators to survive feedwater transients and load rejections, to reduce the frequency of such transients.
- 29% of unplanned scrams during early operation originate in the Main Feedwater (MFW) System.

- Specifically, for both new and mature plants, equipment problems in the MFW System originate most often with the MFW pumps, control valves and their associated control systems.
- Equipment problems accounted for 52% of all unplanned MFW-initiated scrams while human error accounted for 31% of such scrams. These scrams occur most often while attempting to change plant power level (36%), followed by steady state operation (23%) and testing (19%).
- The most prevalent MFW problems involved:
  - — flow control during startup and at low power levels.
  - — regulating valve control systems.
  - — pump control systems.
- The turbine system causes about 15% of all scrams for new plants, and 10% for mature plants.
- Specifically, the turbine, Main Steam and steam dump (or bypass) systems account for roughly 18% of the new plant scrams.
- Equipment failure accounts for 56% and human error accounts for 24% of such Turbine/Main Steam-initiated scrams. Testing contributes in 46% of such cases, followed by 26% during power changes. Turbine stop valve testing and MSIV surveillance testing are largely responsible for the testing contribution.

2.7.2 Operating Experience Feedback Report – Progress in Scram Reduction (NUREG-1275, Volume 5)[14,15]

As a followup to the previous Operating Experience Feedback Report described in 2.7.1 above, AEOD issued Volume 5 of NUREG-1275 in March 1989. As in the previous report, new plants are defined as those plants which had received their operating licenses within the past 2 years. There were 34 different plants which were within two years of receiving an operating license for at least part of the time between January 1, 1984 and December 31, 1987, the study period. A total of 1885 unplanned reactor scrams occurred during the four-year period 1984-1987. Of these, 1319 occurred at plants with more than two years of operational experience and 566 were experienced by plants with two years or less of operational experience. A "scram" is defined as any RPS actuation, either automatic or manual, that resulted in control rod motion. RPS actuations without control rod motion, which incidentally occur in large numbers while the reactor is shut down, are not included. If a given plant was in its first two licensed years for any part of the period from 1984 through 1987, scrams experienced during that period were pooled, averaged, and analyzed as "New Plant" experience. For the same plant, scrams which occurred after completion of two years of licensed operation were analyzed as part of the "Mature Plant" experience. Thus, plants were phased into the mature plant statistics over the period analyzed.

The results are provided on both a generic overall basis as well as by each NSSS vendor, and by new plant versus mature plant. With respect to BOP systems, the following is noted:

- In general, scrams at new plants arise from the same sources as scrams at mature plants, i.e. the Main Feedwater, Main Turbine and Reactor Protection Systems. The new plants experienced scrams attributable to those systems at a higher rate than mature plants.
- The power conversion systems, i.e. Feedwater, Main Turbine, Main Generator, Main Steam and Condensate Systems are the dominant causes of unplanned reactor scrams at new plants.
- At new plants, typical Feedwater System equipment malfunctions originated in piece parts, such as printed circuit cards, switches and controllers, that in turn led to malfunctions of feedwater pumps and of system valves, e.g.

feedwater regulating valves, feedwater system isolation valves, and feedwater bypass valves.

- For mature plants, the power conversion systems accounted for 58% of the unplanned scrams for the entire period of 1984-1987.
- Similarly, support systems accounted for 17% of the 1984-1987 unplanned scrams at mature plants, with electrical distribution, i.e. 120V AC, switchyard, large plant loads, DC power and control centers, accounting for 13% of the 17% figure. (The remaining 26% are attributable to the NSSS.)
- Equipment failures related to feedwater pumps, which typically occur at higher power levels and primarily to the turbine-driven designs, and feedwater regulating valves were caused by the same kinds of faults across the industry (problems with valve operators, controller cards, control oil, and lube oil).
- Specifically for feedwater valves, a significant number of valve operator failures caused by system- or valve-induced vibration have occurred, as well as by oil, moisture, and/or rust or foreign particles in the instrument air system. Contained leakage problems resulting from damaged valve trims (plug and cage or sets) and improperly adjusted valve operators have resulted in steam generator overfill and subsequent reactor trips. Corrective measures have been identified for all of these problems.
- For motor-driven feedwater pumps, premature bearing and seal failures have occurred due to poor pump-to-motor alignment, thrusting or pump instability problems, excessive pump vibration, water contamination of the lube oil, intrusion of foreign material such as dirt or particles into the seal water, seal design weaknesses and poor installation. Shaft or impeller failures have occurred due to cyclic fatigue from normal to excessive vibration or other

loadings. Casing erosion/corrosion problems due to long-term wear or corrosion have also been observed. Corrective measures have also been identified for all of these problems.

- For turbine-driven feedwater pumps, lube and control oil contamination has been caused by water getting into lube oil from steam leaking past turbine steam and lube oil seals. Lube and control oil leaks have been caused by broken or cracked hoses or piping and gasket problems. Governor or control failures have occurred due to dirty control oil, oil leaks, mechanical failure or aging. Premature turbine, pump bearing, or rotating element failures have been caused by improper alignment and/or rotating element balancing. Pump or turbine trips due to low oil pressure have been caused by sudden decreases in lube oil pressure during pump switchover. As before, corrective measures have been identified.
- The main turbine system was the second leading contributor to unplanned scrams, at an essentially constant rate during 1984-1987. Of these, turbine surveillance testing caused the largest number of scrams. Equipment failure-initiated scrams involved switches, relays, circuit cards, fuses and voltage regulators. Main turbine lubrication subsystem problems generally involved failures, strainers, unloaders, motor operated valves, and piping. Turbine blade failure did occur but was rare.
- One half of all main turbine equipment-initiated scrams came from electro-hydraulic control (EHC) subsystems. The electronic failures consisted mainly of circuit cards relays and switches. Hydraulic system failures involved contaminated (dirty) oil, sticking valves, fluid leaks and pressure losses.

- The main generator failed primarily by loss of excitation voltage, leading to loss of generator load, turbine trip, and unplanned reactor scram. This is an example of a single point failure that invariably led to a scram.
- For support system-initiated scrams, the electrical distribution system was the dominant contributor. Other contributing systems included the circulating water, service/instrument air, fire protection and HVAC systems. The 120V AC instrument system was the dominant contributor to electric distribution system-initiated scrams. Equipment failure was the dominant cause of electrical distribution-initiated scrams. Failure often occurred off-site and involved either transmission lines or substation equipment. They usually occurred during steady state operation at or near full power. Generally, no other contributing activity such as testing or maintenance was in progress. Maintenance errors such as operation of the wrong breaker, or improper testing or trouble shooting, were the dominant human error in the electrical distribution system.

2.7.3 Development of Maintenance Effectiveness Indicator

AEOD has sponsored a program under FIN L-1345 for the development of a Maintenance Effectiveness Indicator and a preliminary report was issued in October 1989.[16] The study was conducted by analyzing 3881 Nuclear Plant Reliability Data Systems (NPRDS)[17] component failure data for preselected systems and signals any increase in the failure rate that exceeds a predetermined value. The number and frequency of the flagged failure rate increases is then trended for all systems considered over the study period to obtain a measure of the level of maintenance effectiveness at a plant. Based on a review of the NPRDS narrative descriptions, the cause of each failure was assigned to one of five distinct categories. The categories were analyzed to assess the relative contribution of ineffective maintenance to equipment failures. The five categories are Ineffective Maintenance, Random, Design/Installation/Construction, Normal Aging/Wearout/End of Life, and Unknown.

In the report, reference is made to AEOD/S804B[18] in which it is noted that NPRDS does not currently include certain BOP systems and components that have historically been significant contributors to plant outages, such as the turbine-generator and associated support systems, the condenser, the circulating water system, non-nuclear portions of the service water and closed cooling water systems, the instrument air system, and the service air system. In December 1988, May 1989, and June 1989, official steps were taken by the NPRDS User's Group to include the main generator, main turbine, and condenser in the NPRDS reporting scope. Thus, for the purposes of the current study, it does not appear that the Maintenance Effectiveness Indicator can provide any significant input to studying the effects of aging on BOP systems at this time. However, as the program develops further, with the accumulation of data on BOP systems one would expect this program to provide valuable insights into the aging of BOP systems.

2.8 NRR Survey of Plant Design Strengths and Weaknesses

On September 20, 1988, an internal NRC memorandum[19,20,21] was issued from Mr. Dennis M. Crutchfield, Acting Associate Director for Projects, NRR, to all Project Managers (PM) concerning the strengths and weaknesses of plant design. Each PM was asked to identify the strengths and weaknesses of plant hardware and design, separate from operational performance. As part of the guidance given to the PMs, the memorandum incorporates two tables, Table 1 entitled "Reactor Safety Functions and Risk Significant Systems for PWRS" and similarly Table 2, "Reactor Safety Functions and Risk Significant Systems for BWRs."

Of the PWR systems or components identified as risk significant in Table 1, only the Main Feedwater, Offsite AC Power, Fire Protection and Accident Monitoring Instrumentation Systems can be considered to be BOP systems, and all are nonsafety-related, except for the latter. Only the Main Feedwater and Offsite AC Power Systems fall within the minimum set of systems of components that should be considered in the PM's review.

For BWRs, the Feedwater System, the Steam By-Pass Capacity, Offsite AC Power, Fire Protection, and the Accident Monitoring Instrumentation Systems are the only BOP systems listed as risk significant in Table 2, and with the exception of the Accident Monitoring Instrumentation, all are nonsafety-related.

3. IDENTIFYING IMPORTANT BOP SYSTEMS

In any commercial nuclear power plant, there are numerous BOP systems besides the commonly thought of main steam, feedwater, turbine-generator, condensate, etc. The key objective of this initial phase of the study is to identify and prioritize only the most important BOP systems, and consequently to focus the aging study only on those systems. The following methods are potential means to identify important BOP systems.

3.1 Assessment of Important BOP Systems From Prior NRC-Sponsored Activities

In Section 2 of this report, a summary of the NRC-Sponsored BOP activities which have occurred was presented. From nearly all of the references cited, a clear picture emerges that the most significant contributors to unplanned reactor scrams caused by BOP systems are the power conversion systems, i.e. the feedwater, main turbine, main generator, main steam (usually the steam bypass to the main condenser) and the condensate systems. Other significant BOP systems are support systems such as the Electrical Distribution System, and less significantly the Circulating Water, Service/Instrument Air, Fire protection and HVAC Systems. The Electrical Distribution System consists of the 120V AC Power, switchyard, large plant loads, DC power and control centers. At a component level, the feedwater regulating valves and turbine-driven feedwater pumps are significant, as well as the main turbine electro-hydraulic (EHC) control subsystem. Failures in the main generators are also important.

In reviewing the NRC-sponsored BOP activities described in Section 2, the preponderant safety criterion against which important BOP systems are identified is in reduction in challenges to safety systems. Typically this has been represented by reduction in unplanned reactor scrams.[4,5,13,14,15] According to 10CFR50.73, licensees are required to report several other plant conditions besides those causing an unplanned reactor scram, among which are Engineered Safety Feature (ESF) Actuation Signals, Technical Specification violations, deviations or required shutdowns, loss of system safety function, or excessive airborne radioactivity or liquid effluent release.

ESF Actuation Signals and Technical Specification violations or loss of system safety function have been studied.[2,3,13] Generally speaking, the data show that, except for certain BOP systems which may be nonsafety-related but subject to Technical Specification requirements such as Fire Protection or Radiation Monitoring, BOP systems are not significant contributors to those challenges to safety systems.

It should be noted that many more incidents occur at nuclear plants than are required to be reported as LERs. If one were to examine the individual plant records of these less serious incidents, additional or different BOP systems might be identified as showing a high failure rate and whose continued poor operating performance might eventually cause additional challenges to safety systems.

3.2 Categorization Based Upon Probabilistic Risk Assessment (PRA) Insights

As just mentioned, identification of important BOP systems within the NRC-sponsored BOP activities has relied primarily upon measuring against contributions to unplanned reactor scrams. The focus there is on prevention of accidents, a clearly beneficial result.

In the field of PRA, obviously a more global approach is taken. A Level I PRA, in which the overall core damage frequency is calculated, goes well beyond the initial challenge to safety systems represented by the unplanned reactor scram. It must include subsequent loss of decay heat removal capability, either short or long term, represented by failures of mitigating systems. BOP systems such as feedwater, main steam, steam bypass to the main condenser, and condensate normally figure prominently as mitigating systems, as well as means of preventing a reactor scram. During normal plant operation, those systems are serving their intended purpose of removing fission product heat and converting it to electricity via the turbine-generator. As already noted, those systems are prime contributors to unplanned reactor scrams. However, subsequent to a scram, those systems are the preferred means of decay heat removal by means of the steam bypass to the main condenser. Thus loss of those systems would normally force a reactor scram and a very near term reliance on the standby, or safety-related, decay heat removal systems, as well as high or low pressure safety injection, etc.

In a Level II PRA, since the probability of containment failure is the calculated end result, containment heat removal and pressure reducing systems are incorporated into the analysis. These are all safety-related systems and thus do not involve the BOP systems. In the final type of PRA, a Level III, the probability and extent of early and latent health effects to the surrounding population, assuming containment failure, are calculated. Thus no additional insights regarding BOP systems would normally ensue.

In most PRAs, the number of systems modeled in detail by means of fault trees is generally about twelve to fifteen, and typically these are safety-related non-BOP systems. In any given commercial nuclear power plant, the total number of mechanical and electrical systems is usually on the order of three to four times that number.

Since the BOP systems already identified as important with respect to causing unplanned reactor scrams are only a small fraction of the total number of plant systems, an expert opinion survey was conducted using BNL staff familiar with both PRA and power plant design in order to determine which systems they would identify as important with respect to the more global perspective of PRA, i.e. not simply limited to unplanned reactor scrams. The survey was divided between BWRs and PWRs, with the system titles described in most cases according to the Sequence Coding and Search System (SCSS) code listings for systems.[22] From the master list of systems, the individual expert placed the various systems into the appropriate columns, 1 through 6. The objective was to sort all of the plant systems into a logical framework.

Expert opinion was particularly required in assigning systems to Columns 3 and 4. In these columns, systems were located based on whether or not the expert judged them to be potentially important, even though they are not normally included in a typical PRA.

For both BWRs and PWRs (see Tables 3-1 and 3-2), the nonsafety-related BOP (non-NSSS) systems which were identified as being typically included in PRAs, i.e. Column 2, coincided very closely with the BOP systems identified as important in causing unplanned reactor scrams. That is, the typically included systems were the electrical systems such as Normal AC, High Voltage AC, Medium Voltage AC and the power conversion systems such as Feedwater and Condensate, Turbine (or Steam) Bypass, Main Condenser, etc. The Circulating Water and Station Service Water Systems were also identified as appearing in some PRAs.

There were a substantial number of systems not normally included in PRAs which were considered to be potentially important, column 3, such as miscellaneous electrical systems, the turbine building closed cooling water (BWRs) or service water (PWRs), the cooling towers, miscellaneous HVAC systems, monitoring, detection and control systems and other systems such as security and communications. Even without the nonsafety-related NSSS, Column 6, included as BOP systems, the total number of BOP systems is extensive, Columns 2, 3 and 4, yet the vast majority of these systems do not appear as significant contributors to unplanned reactor scrams.

It should not be too surprising that the BOP systems typically included in PRAs coincide quite closely with the previously mentioned important BOP systems according to unplanned reactor scrams. The PRAs by definition must accurately reflect and emphasize the frequency of initiating events, and so their data sources are essentially the same as the LER data base, although plant-specific failures below the severity level of an LER issuance may be used as well.

In a separate, unrelated effort, BNL staff members developed a study of initiating events with special emphasis on reactor trips.[22] This effort was part of a program to develop risk-based performance indicators (RBFI). The study notes that in PRAs, reactor trips are categorized into four initiating type events:

TABLE 3-1 TYPICAL BOILING WATER REACTOR (BWR) SYSTEMS IMPORTANCE CATEGORIZATION

NON-NSSS				NSSS	
	NON-SAFETY-RELATED				
(1) Safety-Related	(2) Typically included in PRAs	(3) Not in PRA (Potentially Important)	(4) Not in PRA (Not Important)	(5) Safety-Related	(6) Non-Safety-Related
<u>Electrical Systems</u> - Norm. Aux. AC Power (NAACP) - Standby AC Power (SACP) - Essen. AC Power (EACP) - DC Power (DCP) - Switchgear/Load Centers/MCCs - Power/Control Cables <u>Cooling Water Systems</u> - Standby/Emerg. Service Water - RHR Service Water - Safety Rel. Chilled Water (e.g., RBSVS, CRAC) - Fuel Pool Cooling & Cleanup <u>Heating, Vent., & Air Cond. (HVAC) Systems</u> - Rx. Bldg. Standby Vent. (RBSVS) - Control Room Air Cond. (CRAC) - Sec. Cont. (Standby Gas Treatment) - Drywell/Torus - Rx. Aux. Bldg. - EDG Bldg.	<u>Electrical Systems</u> - Normal AC - Turbogenerator - High Voltage AC[1] - Med. Voltage AC[1] <u>Cooling Water Systems</u> - Component Cooling Water (CCW)[1,2] - Circulating Water[4] - Station Service Water[4] <u>Fire Protection[4]</u> <u>Power Conversion System</u> - Main Steam - Cond. & FW - Turbine Bypass	<u>Electrical Systems</u> - Station Transformers (NST & RST) - Isolated Phase Bus - Non Seq. Busses - Substations - AC Instrument power - Main Power Transfer - Emergency Lighting - Elect. Heat Tracing - Lighting & Taxed Motive Power - Grounding & Cathodic Protection <u>Cooling Water Systems</u> - Circulating Water[4] - Turb. Bldg. Closed Loop Cooling Water - Reactor Enc. Cooling Water - Screenwell Canals - Cooling Tower System <u>Communication System</u> <u>Misc. HVAC Systems</u> - Primary Containment Atmos. Control - Aux. Equipment - Rx. Bldg. Normal Ventilation - Main Chilled Water - Drywell Chilled Water	<u>Misc. Electrical</u> - Grounding System - Unit Prot. & Metering - Lighting (yard/bldgs.) - Cathodic Protection <u>Vents & Drains</u> - Plant Buildings - Equipment - Roof <u>Domestic Water/Saint. Sewage</u> <u>Radwaste</u> - Gaseous - Liquid - Solid - Decont. System - Steam Seal & Radwaste Steam <u>HVAC</u> - Fuel building - Waste Manage. bldg. - Turbine bldg. <u>Test & Servicing Aids</u> - Storage - Turning Gear - Vac. Priming & Air Removal - Sample System - Primary Cont. Int. Leak Rate Test - Chlorination System - Cooling Tower Blowdown	<u>Reactor Vessel and Internals</u> <u>Fuel and Control Rod</u> <u>Control Rod Drive (CRD)[1]</u> <u>Reactor Vessel Instrumentation</u> - Rx Water Level Control - FW Control <u>Containment Systems</u> - Primary Containment (Suppression Pool) - Secondary Containment <u>Nuclear Steam Supply Systems (NSSSS)</u> - MSIVs[1] <u>Neutron Monitoring[3]</u> - Intermediate Range Monitoring - Local Power Range Monitoring - Average Power Range Monitoring <u>Reactivity Control</u> - Reactor Protection (RPS) - Standby Liquid Control (SLC) - Alt. Rod Injection (for ATWS)	<u>Recirculation[1]</u> <u>Main Steam</u> <u>Condensate and Feedwater</u> <u>Reactor Core Isolation Cooling (RCIC)</u> <u>Reactor Water Cleanup (RWCU)</u> <u>Process Instrumentation and Control</u> - Electro Hydraulic Control (EHC) - Feedwater Control <u>Standby Gas Treatment</u> <u>Neutron Monitoring</u> - Source Range Monitoring - Traversing Incore Probe <u>Computer System (Process)</u> - Performance Monitoring - Display <u>Reactivity Control</u> - Rod Control & Info. - Recirc. Flow Control - Reactor Manual Control (RMC) - Rod Sequence Control System (RSCS) - Rod Worth Minimizer (RWM) - Rod Block Monitor (RBM)

NON-NSSS				NSSS	
	NON-SAFETY-RELATED				
(1) Safety-Related	(2) Typically included in PRAs	(3) Not in PRA (Potentially Important)	(4) Not in PRA (Not Important)	(5) Safety-Related	(6) Non-Safety-Related
<u>Structures & Buildings</u> - Primary Contain. - Secondary Contain. <u>Emergency Generator (Diesel)</u> - Lube Oil - Fuel - Starting - Cooling - I&C <u>Compressed Air/Gas Systems</u>[1,2] - Instrument Air - Prim. Contain. Inst. Gas <u>Containment</u> - Isolation - Leakage Control - Spray - Press. Supp. Makeup - Comb. Gas Control - Vacuum Relief - Cont. Heat Removal (CHR) <u>Main Steam</u> - MSIV Leakage Control		<u>Monitoring Systems</u> - Bearing Temp. & Vibration - Radiation Area - Process Radiation - Environment - Plant - Leak <u>Service & Handling</u> - Fuel - Rx Vessel - Refueling - In Vessel & Under Vessel - Startup Equipment <u>Detection & Control System</u> - Leak Detection - Fire Detection - Post-Accident Sampling - FW Chemistry Control <u>Structures, Panels & Storages</u> - Main CR Panels - Local Panels & Racks - Aux. Control Panels - Inst. Piping - Screenwell/Canals - Cooling Tower - Fuel Storage - Conduit & Cable Trays	- <u>Plant Stack</u> <u>Matl. & Equipment Handling</u> <u>Seal Water</u> <u>Aux. Steam</u>	<u>Emergency Core Cooling Systems (ECCS)</u> - High Pressure Core Spray (HPCS) - Low Pressure Core Spray (LPCS) - Aux. Depress. System (ADS) - Res. Heat Removal (RHR - LPCI) - High Press. Core Inj. (HPCI) - Cond. Storage Tank (CST) - Isolation Condenser <u>Nuclear Boiler Overpressure Protection (SRVs)</u> <u>ESF Actuation</u>	<u>Turbine Generator</u> - Main Turbine - Main Generator - Inst. & Control

NON-NSSS				NSSS	
	NON-SAFETY-RELATED				
1 Safety-Related	2 Typically included in PRAs	3 Not in PRA (Potentially Important)	4 Not in PRA (Not Important)	5 Safety-Related	6 Non-Safety-Related
		<u>Generator Support System</u> - Lube Oil (Gen. - Turb.) - Stator Cooling - Hydrogen Seal - CO_2 Purge - Excitation - I&C <u>Power Conversion Systems (PCS)</u> - Moist. Sep. & Reheat. Drains - Turbine Steam Sealing - Steam Extraction <u>Radwaste Offgas</u> - Offgas recombiner - Noncondensible Gas Extraction - Radwaste Heading <u>Compressed Air</u>[1,2] <u>Security</u> <u>Remote Shutdown Panel</u>[4] <u>Makeup & Purification</u> - Demin. & Makeup Water - Cond. Demineralizes - Cond. Transfer			

<u>Notes:</u>

1 Partially safety-related.
2 Safety-related at some plants.
3 Reactor protection - safety related; Monitoring portion - non-safety related.
4 Included in some PRAs.

TABLE 3-2 TYPICAL PRESSURIZED WATER REACTOR (PWR) SYSTEMS IMPORTANCE CATEGORIZATION

NON-NSSS				NSSS	
	NON-SAFETY-RELATED				
1 Safety-Related	2 Typically Included in PRAs	3 Not in PRA (Potentially Important)	4 Not in PRA (Not Important)	5 Safety-Related	6 Non-Safety-Related
<u>Electrical Systems</u> - Norm. Aux. AC Power (NAACP) - Standby AC Power (SACP) - Essen. AC Power (EACP) - DC Power (DCP) - Switchgear/Load Centers/MCCs - Power/Control Cables/AC Vital Power <u>Cooling Water Systems</u> - Standby/Emerg. Service Water - Component Cooling Water - Fuel Pool Cooling & Cleanup - Safety Rel. Chilled Water (e.g., CRAC) <u>Heating, Vent., & Air Cond. (HVAC) Systems</u> - Rx. Bldg. HVAC - Rx. Aux. Bldg. HVAC - Control Bldg. HVAC - Emerg. Gen. Bldg. HVAC <u>Structures & Buildings</u> - Primary Contain. - Secondary Contain.	<u>Electrical Systems</u> - Normal AC - Turbogenerator - High Voltage AC[1] - Med. Voltage AC[1] <u>Cooling Water Systems</u> - Circulating Water[4] - Station Service Water[4] <u>Power Conversion System</u> - Main Steam - Condensate - Feedwater - Feedwater Control - Turbine Bypass - Main Condenser - Non-nuclear Instrumentation <u>Fire Protection</u>	<u>Electrical Systems</u> - Station Transformers (NST & RST) - Isolated Phase Bus - Non Seq. Busses - Substations - AC Instrument Power - Main Power Transfer - Electric Heat Tracing - Emerg. Lighting & Taxed Motive Power - Grounding & Cathodic Protection - Conduit & Cable Trays <u>Cooling Water Systems</u> - Turb. Bldg. Service Water - Cooling Tower - Chilled Water <u>Power Conversion System</u> - Turbogenerator - Turbogenerator Steam Sealing - Turbogenerator (I&C) <u>Power Conversion System</u> - Turbine - Steam Sealing - Lube Oil - I&C	<u>Misc. Electrical</u> - Grounding System - Unit Prot. & Metering - Lighting (Yard/Bldgs.) - Cathodic Protection <u>HVAC</u> - Fuel Building - Waste Management Bldg. - Turbine Building <u>Auxiliary Steam</u> <u>Vents & Drains</u> - Plant Buildings - Equipment - Roof <u>Potable & Sanitary Water</u> <u>Radwaste</u> - Gaseous - Liquid - Solid <u>Miscellaneous</u> - Plant Stack - Material & Equip. Handling - Turning Gear - Primary Containment Integrated Leak Rate Test - Chlorination System - Cooling Tower Blowdown	<u>Reactor</u> - Reactor Vessel - Reactor Core - Fuel & Control Rods - Control Rod Drive - Primary Coolant - Pressurizer - Safety & Relief Valves <u>Instrumentation & Control</u> - Neutron Monitoring[3] -Low Power Range -Intermediate Power Range -Average Power Range - Reactivity Control -Reactor Protection -ATWS Mods. (Alt. Rx Trip) - Solid State Protection/Control - Engineered Safety Features Actuation <u>Emergency Core Cooling/Boration</u> -High Pressure Injection -Intermediate Pressure Injection -Upper Head Injection -Low Pressure Injection/Residual Heat Removal -Borated/Refueling Water Storage Tanks	<u>Reactivity Control</u> - Control Rod Drive - Control Rod Drive Cooling Water - Rod Sequencing - Chemical & Volume Control - Boron Recovery - Neutron Monitoring[3] <u>Process Computer</u> - Performance Monitoring - Display

NON-NSSS				NSSS	
	NON-SAFETY-RELATED				
1 Safety-Related	2 Typically included in PRAs	3 Not in PRA (Potentially Important)	4 Not in PRA (Not Important)	5 Safety-Related	6 Non-Safety-Related
Emergency Generator (Diesel) - Lube Oil - Fuel - Starting - Cooling - I&C Compressed Air/Gas Systems - Instrument Air[1,2] - Prim. Contain. Inst. Gas Containment - Isolation - Leakage Control - Comb. Gas Control - Ice Condenser - Vacuum Relief - Heat Removal Secondary Cooling - Main Steam - Main Steam Pressure Relief - Feedwater - Condensate[1] - Condensate Storage - Auxiliary Feedwater - Steam Generators		- Feedwater Chemistry Control - Condensate Demineralizer - Demineralized Water - Chemical Additive Injection - Non-condensible Gasses Extraction - Steam Generator Blowdown - Generator - Stator Cooling - Lube Oil - Hydrogen Seal - CO_2 Purge - Excitation - I&C - Moisture Separators & Reheater Drains - Extraction Steam Miscellaneous HVAC Systems - Turbine Bldg. HVAC - Pumping Sta. HVAC Monitoring Systems - Environ. Monitoring - Plant Monitoring - Leak Monitoring - Sampling - Fire Detection Compressed Air/Gas Systems - Instrument Air[1,2] - Control/Service Air		- Core Flooding Accumulators - Chemical & Volume Control System/ Emergency Boration Containment - Spray - Isolation Radiation Monitoring	

NON-NSSS				NSSS	
	NON-SAFETY-RELATED				
1 Safety-Related	2 Typically included in PRAs	3 Not in PRA (Potentially Important)	4 Not in PRA (Not Important)	5 Safety-Related	6 Non-Safety-Related
		<u>Miscellaneous Fluids</u> - Insulating Oil - Lube Oil - Fuel Storage <u>Service & Handling</u> - Fuel - Rx Vessel/Head Flange - Refueling - InVessel & UnderVessel - Startup Equipment <u>Structures, Panels</u> - Main Control Room Panels - Local Panels & Racks - Aux. Control Panels - Instrument Piping/Tubing - Conduits & Cable Trays <u>Security</u> <u>Communications</u>			

<u>Notes:</u>

1 Partially safety-related.
2 Safety-related at some plants.
3 Reactor protection - safety related; Monitoring portion - non-safety related.
4 Included in some PRAs.

- Reactor trips with no expected impact on the operability of feedwater and/or power conversion systems (e.g. manual scram, loss of load, etc.).
- Reactor trips with expected impact on the operability of feedwater and power conversion systems (e.g. loss of condenser vacuum, etc.).
- Reactor trip with expected impact on feedwater and power convesion systems as well as the mitigating systems (e.g. loss of offsite power, loss of instrument air, etc.).
- Reactor trips generated by various sizes of loss of coolant events (small, medium, or large LOCAs caused by cascade-failure of RCP seals, inadvertent opening of PORVs, etc.).

In this study, several abnormal events are cited as important indicators which may lead to iniating events such as reactor trip. Other outcomes can be return to normal power operation or normal manual shutdown. It is suggested that the leading RBFIs can be identified by tracking the frequency and duration of these abnormal conditions. The abnormal events cited for either PWRs or BWRs which involve the nonsafety-related BOP systems as defined in this report are shown in Table 3-3.

Table 3-3 Abnormal Events Involving BOP Systems in PWRs and BWRs

PWRs

Failure of Steam Dump to Open
Loss of Instrument Air
Failure of Turbine to Runback Automatically and Manually
Failure of Impulse Pressure Transmitter (Low) (Main Steam Upstream of Turbine Stop Valves
Loss of Condenser Circulating Pump
Loss of One Main Feedwater Pump at High Power
Spontaneous Opening of the Main Generator Output Breakers
Partial Loss of Offsite Power

BWRs

Master Feedwater Controller Failure
Condensate or Condensate Booster Pump Trip
Loss of Feedwater Heater Extraction Steam
Stator Cooling Water Pump Trip (Main Generator)
Steam Jet Air Ejector Malfunction (Main Condenser)
Electrohydraulic Control (EHC) Faults (Main Turbine)

As is evident in Table 3-3, except for the Loss of Instrument Air and Partial Loss of Offsite Power, all of the abnormal events cited involve the power conversion systems such as main steam, feedwater, concensate, main condenser and turbogenerator. The partial loss of offsite power coincides with the electrical distribution system mentioned in Paragraph 3.1 as important with respect to unplanned reactor scrams. This list coincides quite closely with the systems identified previously and thus provides further substantiation of the choice of important BOP sysems.

4. INTERIM RECOMMENDATIONS AND FUTURE EFFORTS

From a comparison of the means to identify important systems, and the respective results, discussed in Section 3, the data show that the power conversion systems are the prime source of unplanned reactor scrams, followed by the electrical distribution systems and the Circulating Water, Service/Instrument Air, Fire Protection and HVAC systems.

The frequency of unplanned reactor scrams as a performance indicator is the most frequently cited measure and the results appear to coincide with the results obtained with the more global approach inherent in the development of a PRA. Failures of the power conversion and electrical distribution systems can appear as both transient initiators and failures of mitigating systems in a PRA, hence they are of particular importance.

In the next phase of this program, several of the oldest plants will be considered as a group and their LERs involving unplanned reactor scrams since the beginning of commercial operation until the present time will be reviewed to determine if there is an increasing frequency of unplanned scrams caused by or involving, the previously cited BOP systems. This group of oldest plants will include the ones proposed by industry as the pilot plants for the plant life extension study, Monticello and Yankee Rowe.

A group of plants of intermediate age, as well as a group of the newest plants, will also be examined in the same manner. All three groups will be analyzed and compared to each other to identify any trends or patterns.

The ultimate goal is to examine the individual system component failure mechanisms to determine whether aging of BOP systems is a significant factor affecting nuclear plant safety. This will be done in the same manner as has been done for safety-related systems within the NPAR Programs, such as the Component Cooling Water and Residual Heat Removal Systems.[23,24] In conducting these phases of the study, primary emphasis will be placed on using the NRC-sponsored Sequence Coding Search System (SCSS)[25] for LERs and also the Nuclear Power Experience (NPE)[26] data base. NPRDS will not yield much useful information because BOP systems (limited to turbine, generator and condenser) were only recently included in the data base.

5. REFERENCES

1. Stone & Webster Engineering Corp., "Summary of IE Bulletin 79-01 Review – Balance of Plant (Inside & Outside Containment) and NSSS (Inside Containment) for Class IE Equipment – North Anna Power Station Units 1 & 2 – Virginia Electric and Power Company", February 25, 1980, (ACN No. 800260731).
2. T.R. Wolf, M.R. Harper, K. Higgins, "Engineered Safety Feature Actuations at Commercial United States Nuclear Power Reactors – January 1 through June 30, 1984", AEOD/P503, August 1985.
3. M.R. Harper, T.R. Wolf, K. Higgins, "Engineered Safety Feature Actuations at Commercial United States Nuclear Power Reactors – July 1 through December 31, 1984", AEOD/P603, August 1986.
4. L. Bell, P. O'Reilly, M. Harper, "Trends and Patterns Report of Unplanned Reactor Trips at U.S. Light Water Reactors in 1984", AEOD/P504, August 1985.
5. L. Bell, "Trends and Patterns Report of Unplanned Reactor Trips at U.S. Light Water Reactors in 1985", AEOD/P602, August 1986.
6. R. Lay et al., "Regulatory Considerations for the Balance-of-Plant", NUREG/CR-4783, MTR-86 W00213, MITRE Corporation, Final Report, October 1986.
7. NRC Memorandum from William J. Dircks, Executive Director for Operations, "Davis-Besse Event – NRC Lessons Learned", November 26, 1985.
8. NRC Policy Issue (Information) for the Commissioners, from Victor Stello, Jr., Executive Director for Operations, "Balance of Plant", SECY-86-349, November 21, 1986.
9. NRC Inspection and Enforcement Manual, Temporary Instruction 2515/83, "Balance of Plant Trial Inspection Program (Feedwater System)", Issue Date February 26, 1987.
10. NRC Inspection Manual, Inspection Procedure 71500, "Balance of Plant Inspection", Issue Date September 30, 1988.